LA-UR-00-5186

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Technical Letter Report
GSI-191 Study: Technical Approach for Risk Assessment of PWR Sump-Screen Blockage

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Technical Letter Report

GSI-191 STUDY: TECHNICAL APPROACH FOR RISK ASSESSMENT OF PWR SUMP-SCREEN BLOCKAGE

September 18, 2000, Revision 1

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Prepared for

US Nuclear Regulatory Commission
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Both pressurized water reactors (PWRs) and boiling water reactors (BWRs) rely on the Emergency Core Cooling System (ECCS) to inject water into the reactor core following a postulated break in the reactor primary piping. This accident, commonly referred to as a loss-of-coolant accident (LOCA), and the ability of the ECCS to provide reliable long-term core cooling following a LOCA form the primary basis for licensing of nuclear power plants by the U.S. Nuclear Regulatory Commission (NRC). The Code of Federal Regulations, 10 CFR 50.46, “Acceptance Criteria for Emergency Core-Cooling Systems for Light-Water Nuclear Power Reactors,” requires that all operating reactors (PWRs and BWRs) be equipped with an ECCS that is designed to meet five criteria. One of those criteria is long-term cooling, a process by which core-decay heat is removed by ECCS recirculation water flow.

Damaged insulation and other debris generated by LOCA jets can impede or prevent the recirculation of water into the core in one of two ways. First, the accumulation of debris on sump screens (or strainers) can increase hydrodynamic resistance and thus reduce the net positive suction head (NPSH) available to the ECCS pumps drawing water from the sump. Reduction in the NPSH may result in ECCS pump cavitation, which in turn may degrade the ECCS’s ability to provide long-term cooling. Second, the accumulation of debris at the sump screen or along flow paths on the containment floor may form dams that prevent or impede the flow of water into the sump. This may ultimately lead to a draw-down of water in the sump, which also can cause failure of ECCS recirculation.

The NRC recently completed a research program to study the potential for loss of the ECCS as a result of debris buildup on BWR suction strainers. Based on the results of that study and on experience gained during several operational events, the NRC requested that licensees evaluate their plants and, if necessary, make changes to prevent any detrimental effects from debris blockage(NRC Bulletin 96-03). All BWRs have since installed suction strainers with larger surface areas to ensure that debris blockage does not prevent or impede operation of the ECCS.

In light of the results from the BWR study, the NRC has opened Generic Safety Issue (GSI) 191, “Assessment of Debris Accumulation on PWR Sump Performance.” The overall objective of the GSI-191 program is to carry out a research program similar in breadth and depth to the BWR study that investigates debris blockage of PWR sumps and determines if there is a need for remedial actions.

The Los Alamos National Laboratory (LANL) Technology and Safety Assessment Division (TSA) is supporting the NRC Office of Nuclear Regulatory Research (RES) in a multiyear, multiphase resolution of GSI-191. The LANL research has four technical objectives.

1. Determine if the transport and accumulation of debris in containment following a LOCA will impede operation of the ECCS.
2. If it is found that debris accumulation will impede ECCS operation in some or all PWRs, develop the technical basis for revising NRC regulations and/or guidance to ensure that debris accumulation in containment will not prevent ECCS operation.
3. Provide NRC technical reviewers with sufficient information on the phenomena involved in debris accumulation and how they affect ECCS operation to facilitate the review of any changes to plants that may be warranted.
4. Support the NRC staff in preparation for and during both public and internal meetings concerning the assessment of the effects of debris accumulation on ECCS operation.

One of the criteria that the NRC will use to judge the significance of debris blockage issues for PWRs is the incremental risk of core damage posed industry-wide by the potential loss of the ECCS. This is a comprehensive metric that requires (1) a review of plant-to-plant variability in sump design, containment layout, and Emergency Operating Procedures (EOPs); (2) a careful analysis of all possible reactor accident progressions and an attendant understanding of the time-dependent thermal hydraulics of nuclear reactor systems; (3) a predictive model of debris generation and transport that is both empirically and computationally based; and (4) a thorough description of head-loss phenomena at the sump screen to determine if ECCS recirculation requirements can be met with LOCA-generated debris present.

This paper continues a series of Technical Letter Reports (TLRs) that document the experimental observations, methodologies, and assumptions that are being developed to address the GSI-191 sump-blockage issue. Key elements of an integrated risk assessment methodology are presented here in a general discussion that emphasizes (1) systems-level event-tree models to capture the necessary details of accident progressions and (2) the interface between explicit systems-level events and implicit debris phenomenology simulations that will be needed to estimate the likelihood of ECCS sump availability during a LOCA. A principal objective of the risk assessment methodology is the ability to examine separately the risk effects of different system-failure criteria arising from either licensing-basis assumptions or design-basis plant responses, including alternative mitigation strategies that may be available to operators under EOPs. In many respects, including a rigorous examination of debris effects and the comparison of alternative failure criteria, this methodology is intended to broaden the scope of a conventional probabilistic risk assessment (PRA).

ACRONYMS

AS	Accident Sequence
BS	Break Set
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CP	Conditional Probability
DP	Debris Phenomena
DS	Debris Set
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
GSI	Generic Safety Issue
IPE	Individual Plant Examination
LAFW	Loss of Auxiliary Feedwater
LANL	Los Alamos National Laboratory
LBB	Leak Before Break
LERF	Large Early Release Frequency
LLOCA	Large-Sized Loss-of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LOSP	Loss of Offsite Power
MLOCA	Medium-Sized Loss-of-Coolant Accident
NEI	Nuclear Electricity Institute
NPSHA	Net Positive Suction Head Available
NPSHR	Net Positive Suction Head Required
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation (Office of)
NSSS	Nuclear Steam Supply Systems
PDC	Plant Design Characteristics
PORV	Pressure-Operated Relief Valves
PRA	Probabilistic Risk Analysis
PRT	Pressurizer Relief Tank
PSC	Plant System conditions
PSLB	Pressurizer Surge-Line Break
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Cooling System
RES	Nuclear Regulatory Research (Office of)
SAR	Safety Analysis Report
SCS	Shutdown Cooling System
SGTR	Steam Generator Tube Rupture
SLOCA	Small-Sized Loss-of-Coolant Accident
SS	Sump State
TLR	Technical Letter Report
TRVFO	Transient with Relief Valve Failed Open
TSA	Technology and Safety Assessment

NOTATION

AS	Accident sequence
BS	Break set
CDF	Core damage frequency {yr^{-1}}
ΔCDF	Incremental change in CDF due to debris effects {yr^{-1}}
$CP_{ECCS\ FAIL\ DEBRIS}$	CP of ECCS failure due to debris effects {fraction}
$CP_{SUMP\ AVAIL\ DEBRIS}$	CP of sump availability given possible debris effects {fraction}
DS	Debris set
F_{Sump}	Frequency that ECCS sump is required {yr^{-1}}
$LERF$	Large early release frequency {yr^{-1}}
$\Delta LERF$	Incremental change in LERF due to debris effects {yr^{-1}}
$NPSHA$	Net positive suction head available {Pa}
$NPSHR$	Net positive suction head required {Pa}
$P_{Fail\ Sump\ Debris}$	Binary probability of sump failure for a completely specified accident sequence {0 or 1}

GSI-191 STUDY: TECHNICAL APPROACH FOR RISK ASSESSMENT OF PWR SUMP-SCREEN BLOCKAGE

1.0. INTRODUCTION

This paper summarizes the general approach developed by the Probabilistic Risk and Hazard Analysis Group (TSA-11) at Los Alamos National Laboratory (LANL) for quantifying the risk from debris-induced loss of the Emergency Core Cooling System (ECCS) recirculation sump in pressurized water reactors (PWRs). The purpose of the analysis was to estimate the effect of loss of the ECCS sump on the core-damage frequency (CDF) and the large early release frequency (LERF) for PWRs. Previous estimates of these metrics have not considered the possibility that insulation debris generated during a loss-of-coolant accident (LOCA) may be transported to the recirculation sump.[1] Potential debris accumulation and degraded sump performance are the principal concerns of Generic Safety Issue (GSI) 191, which this work directly supports.

The risk assessment method discussed here was presented at a Nuclear Regulatory Commission (NRC)-sponsored public meeting on March 22, 2000, at NRC Headquarters in Rockville, Maryland. The present expanded and refined report supercedes the previous draft letter report written in April 2000. Although the methodology is presented here generically, it is being developed with the cooperation of two specific volunteer plants to demonstrate its applicability and practicality.

This assessment examines the industry-wide risk of PWR sump-screen blockage, including both large-dry and ice-condenser containment designs with nuclear steam supply systems (NSSS) originally provided by Westinghouse, Babcock and Wilcox, and Combustion Engineering. The participation of volunteer plants with each of these designs and two different vendors (Westinghouse and Combustion Engineering) helps to ensure that templates of all primary safety systems are built into the risk assessment methodology.

Many sources of information are incorporated into this risk assessment. Probabilistic risk analysis (PRA) information embodied in individual plant examination (IPE) studies will be used to obtain plant-specific information. IPEs for the following plants have been gathered: D.C. Cook, South Texas, Oconee, Byron/Braidwood, Haddam Neck, Salem, Watts Bar, Diablo Canyon, and Indian Point 3. Safety Analysis Reports (SARs) also were used. Selected information from the following SARs has been collected: D.C. Cook, South Texas, Zion, San Onofre, Byron/Braidwood, Surry, and Oconee. If necessary, information from other IPEs and SARs will be collected and used as needed.

Other sources of data that will be used in the evaluation of component and system failures include WASH-1400, existing PRAs, and NUREG/CR-5750; information from emergency operating procedures (EOPs) will be used in evaluating potential operator actions that may reduce the likelihood of sump loss.

This risk assessment method is being developed concurrently with ongoing studies of debris generation, containment-pool transport, and sump-screen head loss. Both deterministic and probabilistic models of these phenomena will be needed to assign the conditional probability of ECCS availability for any given accident sequence. Debris phenomenology will be the subject of several separate technical letter reports that support GSI-191, so the present discussion only defines the anticipated interface between event trees (used in the risk assessment to describe the reactor system response) and the detailed simulation of debris transport and accumulation.

2.0. SCOPE OF RISK ANALYSIS TASK

The risk analysis task discussed in this paper is one part of the overall NRC/LANL program for analyzing the effects of debris blockage on the ECCS sump in PWRs. A previous technical letter report to the NRC titled "Selection of Pressurized Water Reactor Accident Sequences for Evaluation of the Effect of Debris in the Sump" established the groundwork for a risk assessment methodology by examining all identified accident sequences for all

[1] Some existing IPE studies examined the possibility of sump blockage but assigned very low probabilities of failure (<1E-3 per demand).

operating PWR reactor designs. Some general conclusions from this study, which helped to determine the appropriate scope of the risk assessment, are listed below.

- There are no available probabilistic models for debris-induced failure of ECCS recirculation in PWRs. Although some IPEs addressed this eventuality, the failure probability data that they incorporated were not based on mechanistic determinations.
- The PWR sump-screen clogging issue is plant-specific. Any future probabilistic models of debris-induced failure of ECCS recirculation also will have to be plant-specific. A significant number of parametric and uncertainty analyses will have to be incorporated into any future analyses if the intent is to draw conclusions regarding the industry-wide risk significance of this issue.
- The probability and timing of ECCS-recirculation failure are strongly dependent on the LOCA size and discharge location inside containment. They also depend on the assumptions related to systems response. Considerations such as (a) licensing-basis plant response compared with design-basis plant systems response and (b) preferred mitigation strategies compared with alternate mitigation strategies should be addressed explicitly.
- Although, the debris-induced pressure drop across a congested sump screen generally should decrease with the size of the break, it appears that a significant head loss could occur even with medium and small LOCAs (for example, when calcium-silicate and fiber debris are generated in combination).
- The timing of ECCS-recirculation failure for smaller LOCAs allows more time for operator corrective actions. Containment spray actuation following a small-LOCA event plays an important role in the transport of debris to the sump, and at the same time, it affects the timing of ECCS recirculation failure.
- Although the most likely mechanism of debris-induced ECCS recirculation failure involves a pressure drop across the screen, other mechanisms are also possible (such as missile generation and screen penetration by debris, which could cause liquid flow restrictions in the core and pump operability or leak-tightness problems). These other effects are not part of this risk assessment, although it can be modified easily to draw insights regarding the risk significance of such concerns.
- Any future models of debris-induced failure of ECCS recirculation are expected to have large uncertainties associated with them, and uncertainty analyses should be an essential part of the risk assessment. The present study proposes an abbreviated uncertainty analysis, especially as it relates to systems response and PRA data. This approach may not capture important coupling that exists between the phenomenological, systems, and reactor-operator interfaces.

To address the above concerns and to fulfill the NRC risk analysis task assigned to LANL, the following subtasks will be completed.

1. Estimate the frequency of important initiating events that lead to a need for long-term cooling by recirculation.
2. For each volunteer plant, estimate
   - the *CDF* and change in *CDF* (ΔCDF) as a result of debris effects,
   - the *LERF* and change in *LERF* ($\Delta LERF$) as a result of debris effects, and
   - the conditional probability (CP) of ECCS failure as a result of debris effects ($CP_{ECCS\ FAIL\ DEBRIS}$).
3. Perform a sensitivity/parametric analysis for PWR plants in general that
   - captures plant design differences and
   - evaluates the importance of major differences that affect risk.

3.0. TECHNICAL APPROACH FOR EVALUATING RISK

3.1. Selection of Evaluation Tools

The following objectives were selected for the risk evaluation approach.

- Estimate *CDF*, ΔCDF, *LERF*, $\Delta LERF$, and $CP_{ECCS\ FAIL\ DEBRIS}$
- Differentiate among plant designs
- Be able to quantify numerous accident sequences at the systems level
- Be extensible to the component level

- Consider operator mitigation strategies
- Be quantifiable with both licensing-basis assumptions and "most likely" plant response
- Be able to quantify the effect of debris accumulation on the sump screen

Several attributes are desirable in the risk evaluation approach.

- State of the art
- Fast (computerized)
- Flexible, extensible, proven
- Acceptable to NRC
- Easy to understand conceptually
- Inclusive of sensitivity and uncertainty analysis capabilities

Based on the objectives and desirable attributes listed above, we recommend that the SAPHIRE software package be used to evaluate the risk. We recommend that plant-response models be developed using event trees defined at the systems level and that the systems-level models be flexible enough to incorporate traditional PRA data, newer PRA data, and plant-specific data. It is recommended that the evaluation models be extended to the component level using fault trees only as necessary to quantify the frequencies of specific events in the systems-level models.

3.2. Components of the Evaluation Process

The evaluation process (other than debris generation and transport phenomena, which are addressed later) has the following components.

1. Select accident sequences
2. Identify possible mitigation strategies
3. Estimate frequencies of initiating events
4. Account for licensing vs "most likely" plant systems response
5. Account for plant design differences

3.2.1. Selection of Accident Sequences. The criteria used to select specific accident sequences for evaluation include (1) the potential importance of the ECCS sump for mitigating an initiating event and (2) the potential of the accident to generate significant quantities of insulation debris, i.e., whether high-pressure fluids are released to containment. The metric used to measure the importance of the ECCS sump is the frequency with which it is required; specifically,

$$\begin{gathered} F_{sump} \equiv \text{frequency of accident-initiating event} \\ \times \text{ conditional probability sump is required for ECCS recirculation.} \end{gathered}$$

Note that these criteria do not address the likelihood that the sump will be blocked by debris because that is the purpose of the follow-on evaluation of the selected sequences. The application of this metric was documented in an April 30, 1999, letter report to the NRC titled "Selection of Pressurized Water Reactor Accident Sequences for Evaluation of the Effect of Debris in the Sump."

Based on the earlier letter report and follow-on discussions with the NRC, the following accident sequences were selected for evaluation.

1. Loss of Offsite Power Followed by Loss of Auxiliary Feedwater (LOSP/LAFW)
2. Medium Loss-of-Coolant Accident (MLOCA)
3. Small Loss-of-Coolant Accident (SLOCA)
4. Large Loss-of-Coolant Accident (LLOCA)
5. Transient with Pressurizer Relief Valve Failed Open (TRVFO)
6. Pressurizer Surge-Line Break (PSLB)

Sequence 1 (LOSP/LAFW) is a transient that requires the use of "feed and bleed"[2] operator action because both main and auxiliary feedwater are lost. This sequence covers non-LOCA transients that evolve into feed-and-bleed scenarios in which the sump is used for long-term recirculation.

Sequences 2, 3, and 4 are LOCAs of various sizes for which the ECCS sump is required for ECCS recirculation. The timing of switchover from ECCS injection to ECCS recirculation and the specific ECCS recirculation pumps and recirculation flow rates differ among these three LOCAs.

Sequence 5 (TRVFO) is a precursor transient event that transitions into a LOCA. Two such transients were considered: (1) a failed-open pressurizer relief valve and (2) a reactor coolant pump (RCP) seal LOCA. The failed-open relief valve was selected for specific analysis because it is the more likely of the two accident scenarios.

Sequence 6 (PSLB) is a break in the pressurizer surge line. At the request of the NRC, sequence 6 was included because some plants may not meet the NRC criteria to credit leak before break[3] (LBB) for this piping system.

Sequences 1 and 5 share the common feature that venting occurs through a rupture disk of the pressurizer relief tank (PRT), which represents a specific location in the containment that may or not contain insulation. All other sequences discharge high-pressure water and steam at the point of the break. The full range of break locations for these sequences *must* be examined parametrically to account for possible debris generation.

3.2.2. Identification of Possible Mitigation Strategies. In the event of an accident or abnormal operating condition, the reactor operators may choose from several courses of action that reduce the severity of the event. These alternatives carry different likelihoods that the ECCS sump will be required to bring the plant to safe shutdown. To date, the following possible mitigation strategies have been identified based on the Nuclear Electricity Institute (NEI) survey of operating PWRs.

- Refill source of injection water and continue injection
  - Requires borated water
  - May overfill containment
- Depressurize Reactor Coolant System (RCS) and use Shutdown Cooling System (SCS)
  - There are limits on the rate of depressurization/cooldown
- Throttle flow through pumps that pull from sump
  - Counter to the safety philosophy of injecting as much water to the vessel as possible
  - May violate requirement to maintain subcooling margin

All potential mitigation strategies require consideration of the appropriate EOPs. Sources of information for consideration of potential mitigation strategies include the PWR plant survey (completed as part of the overall sump-blockage analysis effort—see the separate technical letter report) and EOPs for selected plants.

3.2.3. Estimation of Initiating-Event Frequencies. Three sources of initiating-event frequencies will be used.

- Licensing-basis (all initiating-event frequencies equally likely)
- "Standard" PRA (from the WASH-1400 Reactor Safety Study through the more recent PRAs performed as part of the IPE program)
- "Newer" risk assessment values (incorporating LBB considerations)

[2]"Feed and bleed" refers to a manual procedure for decay-heat removal where the operator periodically opens pressure-operated relief valves (PORVs) to bleed pressurized water from the RCS and then charges high-pressure safety injection pumps from the ECCS sump to feed water back to the RCS.

[3]Experimental tests have shown that RCS piping is susceptible to gradual degradation mechanisms such as wall and weld thinning that will introduce observable leaks long before a significant break occurs. It is argued that a rigorous ultrasound and optic inspection program can greatly reduce the estimated frequency of catastrophic failure.

The reason for using these different sources of data is to effect risk quantification using assumptions that span the range from most conservative estimates of initiating-event frequencies to the most realistic. Licensing bases examine the frequencies of each severe accident independently without regard to relative frequencies of occurrence. Traditional PRAs apply an equally conservative estimate of each accident initiation frequency and then compare the relative risk contributions of each. Recent considerations of LBB have attempted to replace recognized conservatism in the traditional break-frequency estimates with realism supported by experimental study.

Specific frequency data from these sources were included in the April 30, 1999, letter report to the NRC titled "Selection of Pressurized Water Reactor Accident Sequences for Evaluation of the Effect of Debris in the Sump." Table 1 is an excerpt from that letter report that lists frequency values from various sources.

3.2.4. Accounting for Licensing-Basis vs "Most Likely" Plant Systems Response. For the most part, licensing assumptions reflect a single-failure criterion that results in the availability of only one of two ECCS trains to provide core cooling during both the injection and recirculation phases of an accident and only one train of containment cooling with sprays and (possibly) fan coolers. Regulatory Guide 1.1 typically is applied, which does not credit the effect of containment pressurization on the net positive suction head available[4] (NPSHA) for the ECCS sump.

In contrast to the single-failure licensing criterion, the "most likely" (most probable) response is that all equipment will be available, which results in the operation of both trains of the ECCS and both trains of containment cooling. It is possible that the "most likely" plant response may result in a higher likelihood of sump loss than the licensing response because the "most likely" response requires increased flow from the sump that may increase debris transport and lead to a higher pressure drop across the sump screen. However, if fan coolers prevent the actuation set point for containment spray from being reached, the "most likely" response may result in a lower likelihood of sump loss because less volume will be required from the sump. In any case, the "most likely" response will result in an increased containment pressure that increases the NPSHA; this increase is not credited by the licensing bases. It is difficult to determine the effects of these assumptions on sump availability without a systematic examination of each alternative.

3.2.5. Accounting for Plant Design Differences. There are numerous important differences among the operating plants that affect the availability of the sump when debris is present. Some of the most important of these differences are as follows.

- Sump and pump characteristics
- Use of makeup pumps as part of the high-pressure ECCS
- Use of fan coolers for containment cooling
- Point of discharge of RCS safety valves, i.e., to the pressurizer quench tank or directly to containment
- Different actuation set points for containment spray
- Location of the sump relative to the steam-generator cavities where the largest amounts of debris might be generated
- Types and locations of insulation types used

After risk evaluations for the volunteer plants have been completed, the effect of the variability of important plant design differences will be captured through sensitivity analyses. The final ΔCDF and $\Delta LERF$ values will be expressed with a range that incorporates these differences. This approach for extrapolating specific results from the volunteer plants to the overall population of operating PWRs will address important design differences while minimizing the number of plants that must be modeled in detail.

[4]The inlet pressure head that is available to drive pump performance is determined by the depth of water above the inlet, the water temperature (density), and the pressure of the sealed containment environment among other factors. With the presumption that the summation is greater than 0, this total pressure is defined as the NPSHA. Pump performance improves with increasing NPSHA.

Table 1
Calculations Used to Estimate Frequency that Sump Will Be Required

Debris Concern Category	Accident Condition Type	Accident Condition Frequency (per year)				Conditional Probability of Requiring Sump w/o Special Strategy		F_{sump}, Frequency of Accident Condition Times CP of Requiring Sump (per year); (all sequences potentially generate debris)		Characterization of Potential Source of Debris
		IPE	Basis	Updated	Basis	Value	Basis Notes	IPE	Updated	
A	LLOCA	5E-04	a	5E-06	b	1		5E-04	5E-06	C1
A	MLOCA	1E-03	a	4E-05	b	1		1E-03	4E-05	C2
A	SLOCA (4)	1E-03	a	5E-04	b	1		1E-03	5E-04	C3
A	ISLOCA inside containment	1E-04	c			1		1E-04		C2
A	Transient that transitions to RCP seal LOCA	4E-05	d			1		4E-05		C3
A	Transient involving RCS valves opening and failing to reclose	4E-04	e			1		4E-04		C4
B	Small-small LOCA	1.3E-02	a			1E-03	h, 1, 2	1.3E-05		C4
B	Transients involving RCS valves opening and reclosing	1E-01				1E-03	h, 1, 2	1E-04		C4
B	Transients that discharge fluid into containment but do not evolve into LOCAs (e.g. MSLB, MLFB)	2.16E-03	a			1E-03	h, 1, 2	2.16E-06		C1
B	ATWS transients in which RCS valves reclose	1.3E-04	f			0	3			
C	(None identified at present time)									
D	Transients that do not discharge fluid into containment	8.4	a	1.2	b	1E-03	h	8.4E-03	1.2E-03	C4
D	Steam generator tube rupture (SGTR)	1E-02	a	7E-03	b	1E-03	h	1E-05	7E-06	C4
D	ISLOCA outside containment	2E-06	g			0	3	0		

Basis: (a) Indian Point 3 IPE list of generic values, IPE Table 3.3.1.1; (b) "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987–1995," NUREG/CR-5750, February 1999. (c) based on estimated failure rate of inboard RHR shutdown cooling line isolation valve (see text for additional details); (d) based on estimated frequency of station blackout (SBO) and non-recovery of AC electrical power within 1 h (see text for additional details); (e) based on demand probabilities of PORV operation following a transient, along with probability that an open PORV will fail to reclose (see text for additional details); (f) based on Indian Point 3 IPE estimate of RPS failure probability of 1.6E-05 (see text for additional details); (g) Indian Point 3 IPE list of plant-specific values, IPE Table 3.3.1.1; (h) based on Indian Point 3 IPE loss of secondary cooling (see text for additional details).

Notes: (1) loss of steam generator cooling for decay heat removal, (2) debris from feed and bleed (potential), (3) cannot mitigate with sump, and (4) does not include random RCP seal failures—these failures will be addressed later.

Debris Concern Category: A = some debris/ECCS required, B = some debris/ECCS not required, C = no debris/ECCS required, D = no debris/ECCS not required.

Characterization of potential source of debris: (C1) large (C2) medium (C3) small (C4) debris from feed and bleed; quench tank rupture disk is source of fluid.

3.3. Use of Event Trees with Systems-Level Models

Analyses of event trees with the SAPHIRE computer code are fast. The use of event trees is acceptable to the NRC because event trees have been used extensively for reactor safety modeling since the 1970s. Event trees also are easy to understand conceptually. At the top level, an event tree provides a clear description of systems-level successes and failures through each accident sequence, which are defined by unique paths through the tree. At the top level, an event tree also quantifies each alternative accident sequence by the product of the initiating-event frequency and the subsequent systems-level conditional probabilities at each branch. The outcomes of each sequence quantified in this manner are called "endstates." The SAPHIRE framework has built-in capabilities for conducting sensitivity and uncertainty analyses on each endstate estimate.

A simple example event tree is shown in Fig. 1. The events on the tree are defined at the functional/systems level, and the use of various data sources and of detailed debris phenomena evaluations are both indicated on the figure. The first heading to the left is the initiating event with units of annual frequency (events per year). All other headings define subsequent events (sometimes called "top events") that have unitless probabilities of occurring during any particular accident sequence. Here, each endstate has been given a qualitative evaluation to assess the outcome of each possible accident sequence.

CDF is determined by the states of plant systems that are involved with keeping the reactor core cooled; core damage is postulated to occur when these systems are unavailable or do not function during a particular accident sequence. *LERF* is determined by the state of containment given a release of radionuclides from the core; a significant release is assumed to occur if containment has been breached in any way during a particular accident sequence.

To estimate *LERF* and *CDF*, the event trees will include containment-state information as well as core-cooling-state information in the set of top events for each tree, and the event-tree-sequence endstates will include both core- and containment-state designations. Figure 2 shows how *CDF* and *LERF* will be addressed at a simple conceptual level in the event trees. Event-tree endstates define all outcomes of an accident that are possible under the systems model. For example, the summation of all quantified events ending in core damage approximates the *CDF*.

The event trees will include the quantification of debris effects by using the conditional probabilities for failure of core/containment cooling during recirculation from the sump as explicit events on the trees. For a given accident sequence, this conditional probability is dependent on (a) the state of the plant as specified by the unique set of prior events in the sequence and (b) debris generation and transport phenomena. The first of these considerations is built into the event tree by quantifying the various systems-level branches in the tree. The second consideration will be addressed by using the results of phenomenological debris studies being performed in other tasks of the overall NRC/LANL program.

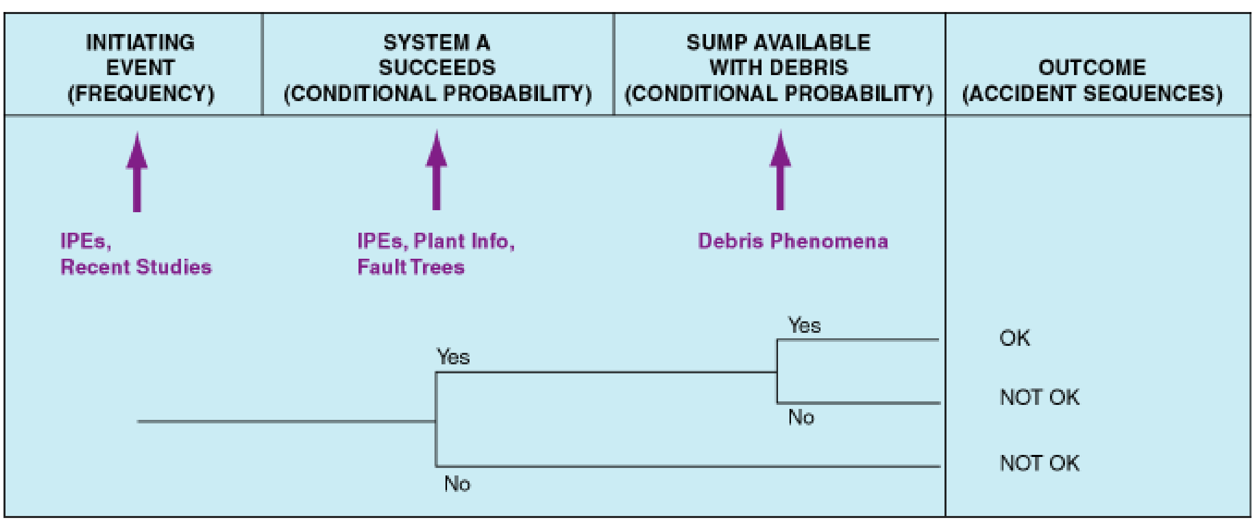

Fig. 1. A simple example event tree.

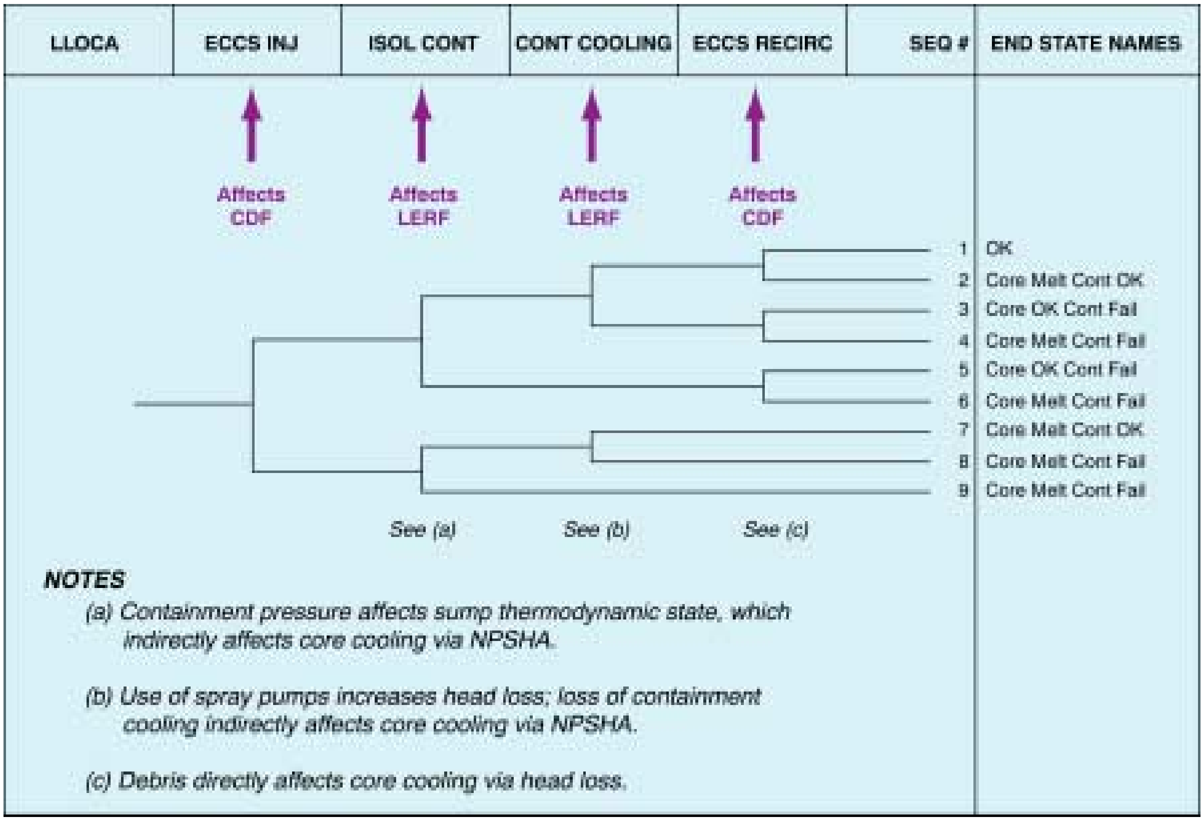

Fig. 2. CDF and LERF included in an event tree.

The event-tree structure defines the plant conditions for any particular accident sequence so that results from the studies of debris phenomena can be used to estimate a conditional probability of sump-recirculation failure. Such plant conditions include the number of pumps using the sump for core/containment cooling, the size and location of the break where fluid is released into containment, the time following the break when the sump is required, and so on. Figure 3 shows how an event explicitly addressing the effects of debris will be included in each event tree.

The event-tree structure also explicitly allows quantification of numerous accident sequences at the systems level. Each initiating event has a unique event tree that delineates all possible accident progressions as combinations of systems successes and failures. Event trees can be extended to the component level by modeling the success probability of each event with a fault tree. Analysis tools such as SAPHIRE automatically link the fault trees for the various events in a sequence. Fault trees may be needed in this study to quantify the success probability of some events for which there are no available industry data.

Possible strategies that operators might use to limit the severity of an accident, such as “feed and bleed” or recirculation-pump throttling, can be explicitly included in the event trees as shown in Fig. 4. Successful mitigation will reduce the likelihood of core damage. The availability of some strategies is highly plant-specific, so these events will be introduced parametrically to help judge their effects on the industry-wide risk of sump blockage.

All accident sequences can be quantified with both licensing-assumption and most-likely success probabilities as shown in Fig. 5. Contrary to what is shown in the figure for the purpose of an example, all events in the tree will be quantified by *either* “most likely” probabilities *or* licensing-basis probabilities for any given analysis.

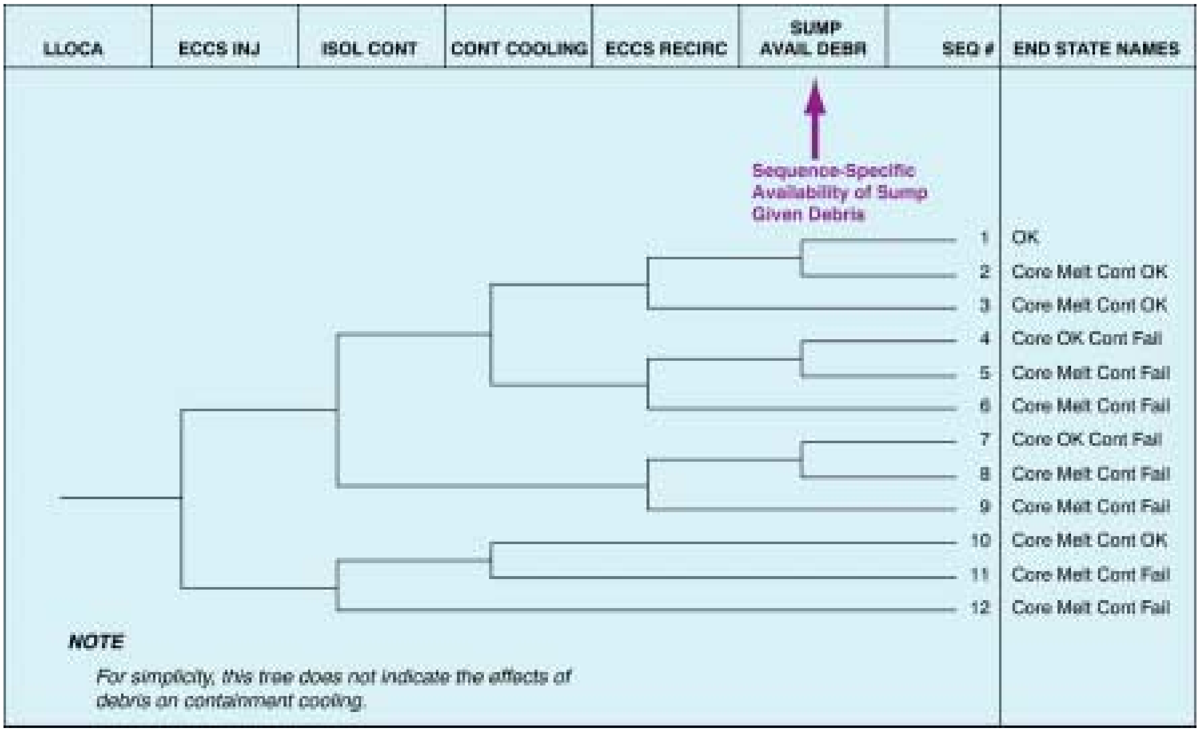

Fig. 3. Explicit inclusion of debris effects in an event tree.

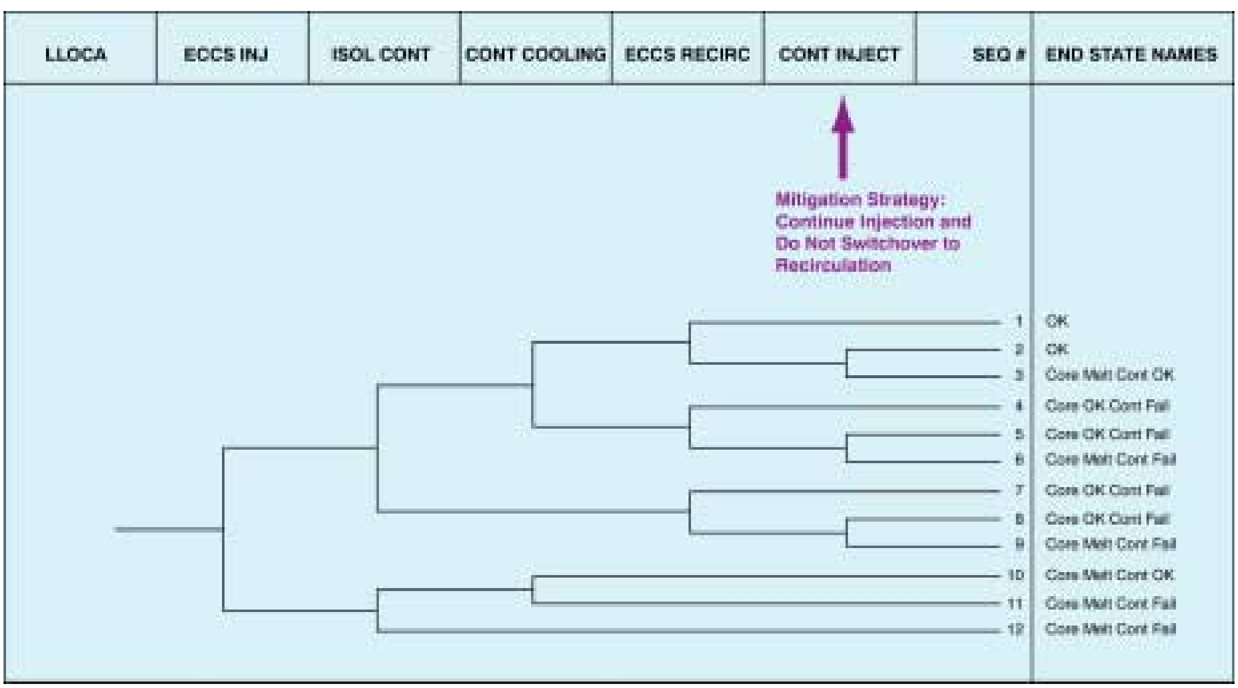

Fig. 4. Explicit consideration of possible mitigation strategies.

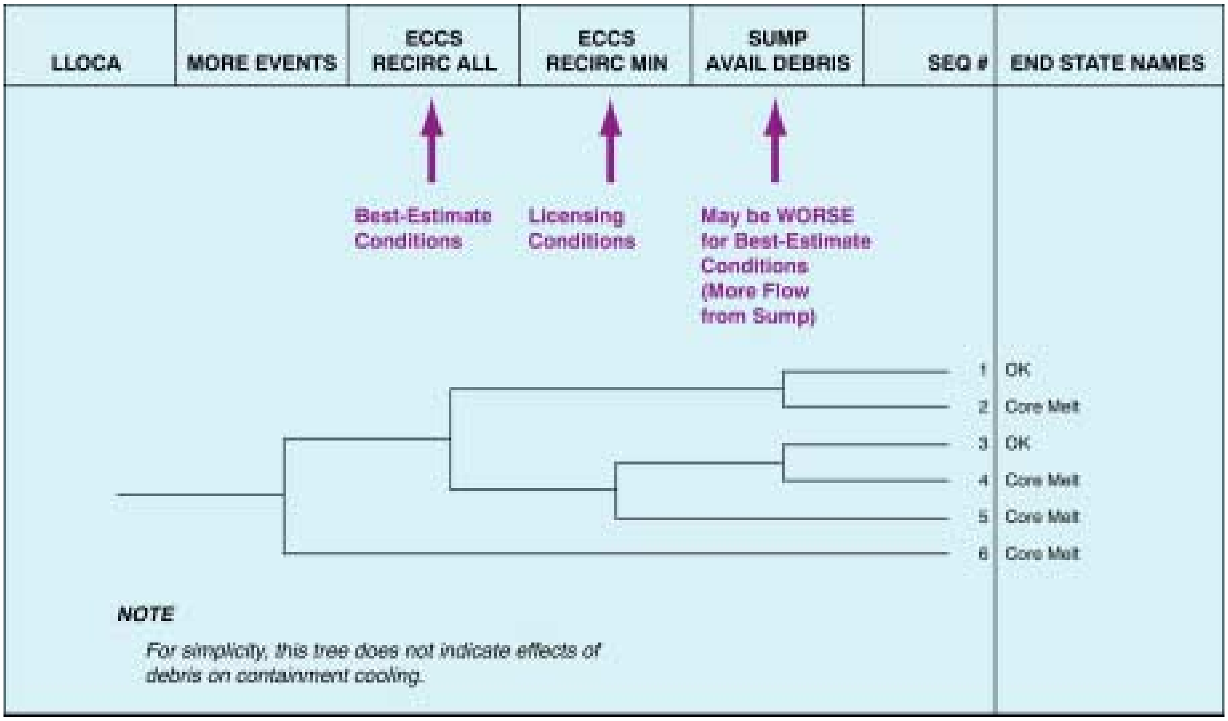

Fig. 5. Consideration of licensing assumptions and most likely plant response.

4.0. INTERFACE WITH DEBRIS PHENOMENOLOGY STUDIES

4.1. Inclusion of Debris Phenomena in Event Trees

Results from the debris phenomena studies will be used to estimate the conditional probability that recirculation from the sump is not available because of the effects of debris generated by the accident. The manner by which the risk model interfaces with the debris studies is described in this section.

Let $CP_{ECCS\ FAIL\ DEBRIS}$ denote the conditional probability that the sump is not available to provide the required recirculation capacity because of the presence of debris. Because debris transport mechanisms may depend on the particular plant conditions, $CP_{ECCS\ FAIL\ DEBRIS}$ is accident-sequence specific. The event tree for each initiating event delineates the possible accident sequences by including

- an initiating-event description including the annual frequency of occurrence,
- subsequent plant-system states and their success/failure probabilities, and
- an explicit event in the tree (SUMP AVAIL DEBRIS) to introduce debris effects.

All ECCS failure mechanisms that are not related to debris effects will be accounted for in a separate event called ECCS RECIRC because previous estimates of the failure probability for this event are available from other sources. When ECCS failure mechanisms are separated in this way, the influences of debris phenomena can be analyzed explicitly.

Figure 3 showed an event tree that includes the event SUMP AVAIL DEBRIS. The conditional probability assigned to the event SUMP AVAIL DEBRIS depends on the accident sequence and the associated debris phenomena. Note that

$$CP_{ECCS\ FAIL\ DEBRIS} = 1 - CP_{SUMP\ AVAIL\ DEBRIS}$$

for each branch in the tree. *If* an accident sequence is developed in sufficient detail that deterministic models can be applied to evaluate all debris generation and transport phenomena, *then* $CP_{ECCS\ FAIL\ DEBRIS}$ is either 0 or 1, depending on the value of NPSHA relative to the success criterion net positive suction head required (NPSHR); i.e.,

$$CP_{ECCS\ FAIL\ DEBRIS} = 0 \text{ if } NPSHA \geq NPSHR$$
$$CP_{ECCS\ FAIL\ DEBRIS} = 1 \text{ if } NPSHA < NPSHR.$$

Note that NPSHR can be defined either by licensing bases, which incorporate a safety margin, or by the minimum pressure head needed to run the ECCS pumps at the capacity required for the accident sequence. In this way, ECCS success criteria also can be defined in terms of both "most-likely" and licensing-basis plant response.

To develop an accident sequence in complete detail, the following conditions must be defined quantitatively.

1. Plant design characteristics (PDCs) that determine spatial locations of insulated pipes and the sump, concrete boundaries limiting LOCA-jet expansion, debris-transport paths, allowed mitigative actions determined by available equipment, etc.
2. Plant system conditions (PSCs) that define the number (and volume) of safety systems drawing from the sump, the status of containment sprays and containment integrity, etc.
3. Sump-state (SS) parameters like water depth, temperature and flow rates that determine NPSHA
4. All debris phenomena (DP) that affect debris generation at the location of a break, transport to the sump, and build-up on the screen

This set of information provides the interface between plant-system models that describe the status and requirements for emergency operation and the debris phenomenology models that describe the physics of the accident progression and the effect of debris on sump availability. Although the assignment of information into these four categories is somewhat arbitrary and interdependencies probably will be found, the process of itemizing necessary information ensures that the coupling between these two pieces of the risk analysis will be complete. In many respects, this interface is perfectly analogous to the definition of plant damage states that provide a transition between traditional Level 1 PRA, which investigates the severity of a reactor core breach, and Level 2 PRA, which propagates a small set of accident conditions to the point of environmental release.

PDCs are the important design features that affect the propagation of an initiating event into the various accident sequences that may follow. For example, at a specific plant, feed and bleed may be accomplished with either safety *or* relief valves, whereas at another plant, relief valves may be required. PDCs also provide some of the conditions under which debris phenomena are to be evaluated for specific accident sequences. For example, at some plants, the safety valves discharge into the pressurizer quench tank, and at other plants, they discharge directly into containment with the possibility of debris generation.

In addition, the physical design of the plant will always have an important influence on the outcomes of debris-generation and transport scenarios. For example, the presence of concrete "doghouses" may limit the expansion of a LOCA jet in the vicinity of a steam generator, and openings from steam-generator cavities will often dictate the path of water flow from a major break to a remote sump located in an outer annulus. The locations and types of insulation are also examples of PDCs that must be included in the physical plant description.

The following PDCs affect the NPSHA and NPSHR, which represent the ECCS performance metric and success criterion, respectively.

- Sump design (elevation, size, strainers, etc.)
- Pump design (elevation, type, NPSHR)
- Containment design (size, floor details)
- Size of water sources (ECCS injection, accumulators, RCS inventory)

Each accident sequence is defined by a specific combination of an initiating event and subsequent system failures and successes; PSCs are represented by that accident-sequence-specific set of combinations. The PSCs for a given accident sequence can affect the importance of the debris in causing loss of the sump. For example, the total number of pumps pulling from the sump (required to support ECCS and containment-spray recirculation) affects the

NPSHA for every pump, and the number of pumps in operation is specified by the PSCs for the applicable event-tree accident sequences. The following PSCs affect NPSHA and NPSHR.

- Initiating event (determines RCS conditions and containment pressure)
- Requirement to use ECCS (initiating event and subsequent system failures, e.g., transient with loss of feedwater requires ECCS for feed and bleed, and operator mitigative actions, e.g., continued injection)
- State of the ECCS (number of pumps, time to switchover to recirculation, high- or low-pressure pumps in sequence)
- Containment isolation successful or failed
- State of containment cooling systems (core spray on, number pumps, fan coolers on, number of coolers)

The SS is the physical state of the fluid in the sump, which affects the ability of the sump to function with debris present. The SS depends on the PDCs and the PSCs. The following SS considerations affect the NPSHA and NPSHR.

- Plant system conditions
- Sump conditions such as
  - amount of water in sump and
  - pressure and temperature (dependent on containment pressure and temperature)

Debris phenomena affect the likelihood that the sump is lost as a result of debris effects. The DP define the minimum set of information needed to describe the physical generation and transport of debris from the occurrence of a break at a specific location to potential build-up at the sump screen. In a generic sense, the DP can be thought of as particular values of the input parameters required by predictive debris generation and transport models that have been (or are being) developed to explain experimental observations of debris behavior. These models eventually may differ in complexity from simple engineering approximations of generated debris volume to detailed computational fluid dynamics (CFD) models of debris transport. The following DP parameters affect NPSHA and NPSHR.

- Initiating event (size of "break," location of break, discharge path to containment)
- Amount, type, and size distribution of debris source
- Transport of debris to and on containment floor
- Settling of debris in containment
- Effect of debris on head loss
- Other parameters (if necessary)

Factors affecting the ECCS performance metric NPSHA are summarized below.

Physical Property	Important Conditions
Sump/Pump Properties	
Thermodynamic state of sump (containment conditions)	SS
Sump size, pump elevation, head loss with no debris	PDCs
Sump water level	PSCs, PDCs
Debris Effects	
Head loss across strainer with debris	DP

Factors affecting the ECCS success criterion, NPSHR follow.

Physical Property	Important Condition
Pump design	PDCs
Flow rate through pump	PSCs
Pump Speed (not of concern for constant speed pumps)	PDCs
Water temperature (may not be significant)	SS

PDCs and the SS can be addressed using calculations from modeling tools such as RELAP and MELCOR. Both depend on the PSCs that are delineated explicitly in the event-tree structures. Therefore, the event trees will incorporate the PDCs and the PSCs from which the SS can be calculated. The resultant PDCs, PSCs, and SS provide the conditions for which the DP are to be evaluated for a given accident sequence. The DP effect on the given accident sequence will be calculated using phenomenological models developed in the concurrent debris investigation of the overall sump-blockage risk assessment.

Given sufficient details at each step, the basic procedure to quantify the failure of the sump for each specific accident sequence is as follows.

- Specify PDCs
- Calculate PSCs and SS
- Estimate DP
- Determine NPSHA and NPSHR

An issue of some importance is whether it is practical to develop event trees in sufficient detail to quantify $CP_{ECCS\ FAIL\ DEBRIS}$ as either 0 or 1. Recall that all input parameters required for deterministic debris-generation and transport models must be specified for a given scenario before a clear decision can be made regarding sump availability. It is argued in the following discussion that although it *is* practical to address PDCs, SS, and PSCs in detail on the event trees, it *is not* practical to address DP in sufficient detail on the event trees. Additional consideration must be given to the interface between the event-tree systems models and the DP phenomenology models.

The accident sequences developed in the event trees will include all of the details required to specify the PDCs and the PSCs. These parameters represent large-scale plant conditions that dictate major branches or decision points along the possible event sequences. The actions taken at each branch to change the plant status (whether by intent or by equipment failure) are discrete and, in most cases, binary events that are easily accommodated by the event-tree logic. At most, 10 to 12 branches will be needed to capture all plant configurations of interest during an accident. Even the numeric values associated with each plant state are limited to a few possible discrete values. For example, maximum recirculation flow rates will be set by the number of cooling systems that are operating in a given scenario.

By comparison, phenomenology models require the values of many continuous variables that are not divided easily into a manageable number of discrete bins. Factors that complicate specifying $CP_{ECCS\ FAIL\ DEBRIS}$ as either 0 or 1 for each accident sequence include the following.

- The fidelity of RELAP/MELCOR calculations affects the determination of the SS, which must be known to compare NPSHA and NPSHR.
- A given LOCA initiating event comprises a large number of different possible break locations, each with different debris-generation potential, and this affects the DP.
- There is uncertainty in the debris phenomena (volume, transport, settling, etc.) that affects the outcome of an accident sequence in terms of sump availability.

The SS results from RELAP/MELCOR calculations can be included either explicitly through additional events in the event-tree accident sequences or implicitly through the assignment of event-failure probabilities. However, the DP are too complicated to address in detail in the event trees. It is not feasible to develop accident sequences in sufficient detail to uniquely specify all debris phenomena; there are too many parameters with continuous ranges of possible values. Therefore, a composite approach will be used to estimate $CP_{ECCS\ FAIL\ DEBRIS}$ that is derived from a statistical combination of parameters that are important for DP. This statistical combination is discussed below.

In summary, the accident sequences on the event trees will be developed in sufficient detail to uniquely specify all parameters needed to calculate $CP_{ECCS\ FAIL\ DEBRIS}$, except for the parameters associated with DP. In other words, only the parameters associated with PDCs, PSCs, and SS will be handled in the event-tree structure. For each accident sequence, DP effects will be calculated using phenomenology submodels, and $CP_{ECCS\ FAIL\ DEBRIS}$ will be estimated as a composite value derived from a statistical combination of important parameters. There are two potential complications with this approach.

1. Dependent effects between DP and plant status may complicate segregation of the accident sequences into separate pieces, and the approach may require refinements as such effects are encountered.
2. The core/containment interface is inherently complicated. Choices for addressing it include (a) model in detail, (b) model with simplifying assumptions, or (c) use conservative assumptions (e.g., Regulatory Guide 1.1).

Item 1 is complicated because there is no general mathematical approach for describing dependencies in an event-tree structure. Complicated dependent effects will be handled on a case-by-case basis. Item 2 is complicated because a rigorous solution of the core/containment response requires a coupled analysis.

4.2. Quantification of Debris Phenomena as a Composite Value

For each accident sequence, there is a defined "hole" size for release of fluid into containment. This size is largely determined by the industry-standard definitions of large, medium, and small LOCA events. Other important variables will be specified by a Break Set (BS). The BS includes {break size, location, pipe size, reactor system, jet geometry, orientation of a directional jet}. Important considerations that are addressed by parameters in the BS include the following.

- Break location. Not all locations have the same frequency of break; e.g., welds, bends, etc.; and insulation types will not be distributed uniformly in containment.
- Pipe size. A total break in a smaller pipe will not have the same depressurization behavior as the same size hole in the sidewall of a larger pipe.
- Reactor system. Some systems may deserve credit for LBB; others may not.

The variables affecting generation of insulation debris and subsequent transport to the sump given a specific BS will be specified by a Debris Set (DS). The DS includes {volume and type generated, reactor system mounted on, initial spatial distribution, volume and type transported, head loss created}. Many background details such as insulation damage pressures and insulation type, thickness, and installation location will be treated as plant configuration information. For example, it may be of interest to replace the specific insulation types of a volunteer plant with another combination that is prevalent in the industry.

Deterministic models will not be available for all steps of the generation and transport analysis. Therefore, many parameters like break location and possibly jet orientation will be sampled randomly from the total range of possibilities. Stochastic evaluation of uncertain parameters implies that a large number of such evaluations may be required, i.e., the BS and DS may have many elements. If each set is imagined as a rectangular matrix, then the values of any row from BS and any row from DS completely define the conditions for a single postulated accident.

4.2.1. Calculation Process.

For each accident sequence (AS), there will be a specific $CP_{ECCS\ FAIL\ DEBRIS}(AS)$. $CP_{ECCS\ FAIL\ DEBRIS}(AS)$ will be calculated as a weighted combination of a set of 0/1 values; each element in the set is $P_{Fail\ Sump\ Debris}(AS, BS, DS) = 0$ or 1, where *BS* is a unique break set and *DS* is a unique debris set. Note that *DS* is dependent on *BS*. Thus,

$$CP_{ECCS\ FAIL\ DEBRIS}(AS) = \sum_{i\ Break\ Sets}\sum_{k\ Debris\ Sets} W_{i,k}\, P_{Fail\ Sump\ Debris}(AS, BS_i, DS_{i,k}) \quad ,$$

where $W_{i,k}$ is a weighting factor and $P_{Fail\ Sump\ Debris}(AS, BS_i, DS_{i,k}) = 0$ or 1.

The use of weighting factors based on the likelihood of occurrence for each element of the BS and DS will produce the arithmetic mean value of the distribution of possible binomial 0/1 outcomes. This is consistent with the selections that will be made for all other system failure probabilities throughout the event trees. Although complete probability distributions can sometimes be defined for each branch, only the mean values propagate multiplicatively through the tree to provide mean values of the endstate frequencies. More complex techniques for sampling branch probabilities will not be used in this analysis.

Note that the notation has been generalized here to account for the possibility of multiple DS for each BS, but the simulations may be run with only a single DS outcome for each element of the break set. In this case, the above equation collapses to a single summation for all practical purposes.

4.2.2. Simple Discussion of the Calculation Process. The previous section provided the mathematical approach for assigning a value to $CP_{ECCS\ FAIL\ DEBRIS}$ for a given accident sequence using the DP. This section provides a simple physical explanation of the process.

The event-tree structure will be developed sufficiently to uniquely specify the PDCs, PSCs, and SS for each accident sequence on each event tree. It is neither practical nor necessary to delineate all the DP on the event trees. It is not practical because there may be many thousands of combinations of values for the BS and DS parameters discussed previously. It is not necessary because the event tree is defined at a systems level. Just as the event tree does not delineate all the ways that a system can fail, it should not delineate all possible debris phenomena. By a similar analogy, because the mean failure probability of a system is calculated from a statistical combination of failures of constituent components, the availability of recirculation from the sump should be calculated from a statistical combination of DP.

Consider a specific event sequence on a LLOCA event tree. The overall frequency of a LLOCA is the frequency of the initiating event (assume 5E-4/yr). However, the effect of the LOCA on generating debris depends on the type of break (double-ended guillotine or crack), the exact size of the break in the LLOCA range, the proximity of the break to insulation, the type of insulation, the location of the affected insulation in containment, etc. Therefore, the overall break frequency of 5E-4/yr must be distributed over numerous specific breaks whose individual frequencies statistically sum to 5E-4/yr. Each individual break must be evaluated to determine whether the resultant debris causes NPSHA to be less than NPSHR or not because different breaks will have different DP.

The weighting factors $W_{i,k}$ are the fractions of the overall initiating-event frequency that result from the specific accident conditions defined in the break and debris sets. For example, if all large breaks are considered equally likely and there are 1000 breaks, the weighting factor is the same for each break, namely, 0.001. The initiating event defines the overall frequency; therefore, the product of the initiating-event frequency and the weighting factor for a constituent break is the frequency of that break (5E-7/yr/break for each constituent LLOCA break if each of 1000 constituent breaks is equally likely). If it is desired to address factors that cause the likelihoods of various breaks to differ, such as the effect of welds and pipe bends, then the weighting factor can be defined nonuniformly among the constituent breaks. At the present time, this possibility is under evaluation. The sum of all weighting factors for a given event sequence must always equal 1.0.

For a given break, the probability that the sump is lost because of debris effects is 0 or 1 depending on the DP for that specific break. The assignment of the probability that the sump is lost for each break will be based on phenomenological evaluations. For 1000 constituent breaks, there will be a set of 1000 values (each 0 or 1) that can be weighted by their relative likelihoods of occurrence (the weighting factors) to obtain the overall probability that the sump is lost because of debris in the accident sequence of concern. This overall probability is $CP_{ECCS\ FAIL\ DEBRIS}$ for a given accident sequence. Separate $CP_{ECCS\ FAIL\ DEBRIS}$ will be calculated in the same manner for each accident sequence in each event tree.

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